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LHD divertor experimental program

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Abstract

The LHD experiment has just begun. A scenario is presented for LHD divertor experiments. It includes development of LHD divertor components particularly efficient pumping system, local island divertor as a closed pumped divertor, simultaneous achievement of H-mode and radiative cooling (SHC operation) as an H-mode approach in the helical device, high temperature divertor plasma operation for enhancement of the energy confinement. © 1999 Elsevier Science B.V. All rights reserved.

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1. Introduction

The National Institute for Fusion Science (NIFS) has constructed a large, superconducting $l=2$ heliotron/torsatron type device, called the Large Helical Device (LHD) [1,2]. The advantage of the heliotron over the tokamak is its inherent steady state operation, i.e., free of current disruption and no need for current drive. Its experiment started in April 1998, aiming at demonstrating the attractiveness of the heliotron type device at more reactor relevant plasma parameters. The initial plasma was obtained by 2nd harmonic heating of ECRH (84 GHz) at magnetic field of 1.5 T. With wall conditioning, the oxygen concentration in the plasma was reduced, thereby leading to a rapid improvement of the plasma parameters. The initial experimental results will be presented elsewhere later this year. We expect that the divertor will play a key role in improving the quality of the LHD helical plasmas [3]. Various innovative divertor concepts and divertor components have been developed for this purpose and described in Section 2. Then a scenario for LHD divertor experiments is described in Section 3.

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2. LHD divertor

2.1. Helical divertor geometry

In LHD, two divertor magnetic geometries, helical and island geometries are to be employed for diverting the outflowing plasma. The helical divertor utilizes a divertor magnetic configuration inherent in heliotron type geometry. It is a fairly complex and three dimensional configuration [3]. Its structure at a poloidal plane (constant ϕ plane) is shown in Fig. 1. In the outer region outside the closed magnetic surface region, several island layers with toroidal mode number of 10 are embedded and at outer radii, the poloidal mode number of the island layer decreases and the size of the island increases. Eventually the layers overlap, resulting in a stochastic field region. Beyond the stochastic region, there exists a region with multiple thin curved layers. Vagueness of the separatrix and the high local rotational transform and high local shear at the edge on the large major radius side of the torus create the thin layer structure [3]. A puncture plot of the field line in Fig. 1 is obtained by tracing with starting points just inside the stochastic region. Field lines from the stochastic region enter these surface layers and after many toroidal circulations, they reach the 'X-point' of the 'separatrix' and then hit the divertor plates. There exist regions without the puncture

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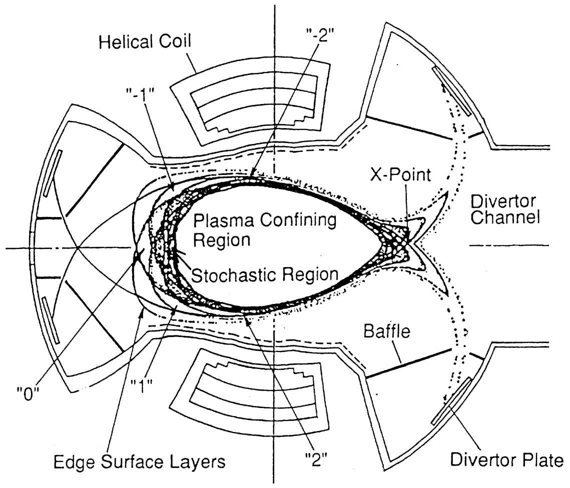

Fig. 1. Schematic view of the LHD helical divertor (poloidal plane at $\phi = 18°$). Edge magnetic structure is shown by a Poincare plot of the field lines. The radial extent of the edge structure for the standard configuration is much smaller than that shown here.

points between the layers, meaning that the field lines in these regions are not connected to the stochastic region. Instead they are connected to the divertor plates. For example, a group of the field lines in region 0 moves as a whole into the region 1 (-1) after 36° toroidal forward (backward) rotation and move into the region 2 (-2) and so on, finally reaching the divertor plate.

2.2. Divertor hardware components

The divertor design needs to be flexible to accommodate a wide range of the divertor operational scenarios. To this end, we designed the largest possible vacuum vessel within the budgetary and technical constraints. The temperature of the vacuum vessel is required to be below 70°C in order to limit the heat flow to the 80 K thermal shield plates which are attached on the back side (i.e., coil facing side) of the vessel by using insulating bolts. The water path of stainless steel with a rectangular cross section (9 mm × 28 mm, 2 mm thick) is welded directly to the vessel wall. The interval of the water paths are 80 mm on the small major radius side of the torus and 200–250 mm on the larger major radius side. The first wall metal plates (1300 mm × 4000 mm) will be installed to protect the vessel from radiative heat flux. Stainless steel and copper plates (with thickness of 5 mm for both metal plates) are attached together to make the first wall plate with high heat transfer efficiency. These plates are designed to handle the radiative heat flux on the vacuum vessel during 3 MW steady state operation.

The LHD divertor target elements are carbon plates fixed to a copper heat sink with a cooling tube. A brazed joint (BJ)-type target element will be the final goal to meet the maximum heat load of 5–10 MW m^{-2}. In the early phase of the experiment, a mechanical-joint (MJ)-type target element will be employed. For MJ target design study, a systematic investigation on the effects of carbon sheets inserted between carbon plates and copper heat sinks was done by varying sheet thickness, materials and torque applied to the bolts. A carbon sheet with thickness of 0.1–0.2 mm is found to be optimum in terms of heat transmission. A simple MJ target system with tolerable steady state heat flux up to 0.4 MW m^{-2} is planned to be installed next year.

The particle control at the edge is very crucial in enhancing the core energy confinement. We are developing two powerful pumping systems for the LHD particle control. In the carbon sheet pumping scheme [4], a large surface near the divertor plates is covered by the carbon sheets, which absorb charge exchange hydrogen particles. Before a series of discharges, the sheets are baked up to 700–800°C to remove the previously trapped hydrogen atoms. After cooling down to below 200°C, the unsaturated sheet can trap charge exchange hydrogen particles. With higher impinging energy, the pumping efficiency of this scheme improve significantly because of (1) higher charge exchange probability over that of the ionization at higher energy, (2) lower reflection probability at higher energy. The optimum thickness of the sheet is a few mm for LHD application. The carbon sheet has a finite pumping capacity of 4×10^{17} cm^{-2} for impinging hydrogen atom energy of 1 keV (it increases with increasing energy of the particle because of deeper penetration into the carbon material at higher energy). Thus the operation time is limited, a few tens of seconds for the LHD. This is the main drawback of this scheme. For a steady state hydrogen pumping scheme [5], we are developing metal membrane pumping, which utilizes superpermeability of niobium. We demonstrated high pumping capacity of up to 3×10^{17} cm^{-2} s^{-1} for hydrogen atoms generated by high temperature (2000°C) filaments [6]. Then pumping of the hydrogen atoms generated by plasma (10^{12} cm^{-3}, 10 eV) in the linear device was demonstrated [7]. More recently, membrane pumping has been demonstrated in the divertor region of the JFT2M tokamak [8]. With successful tests of the membrane pumping so far, we are confident that it will provide very effective pumping in the LHD device even in a steady state fashion.

2.3. Local island divertor (LID)

The LID is a closed divertor utilizing the island geometry [9,10]. The separatrix of the island ($n/m = 1/1$) provides a sharp separation between the closed and open regions. As illustrated in Fig. 2, the outward heat and

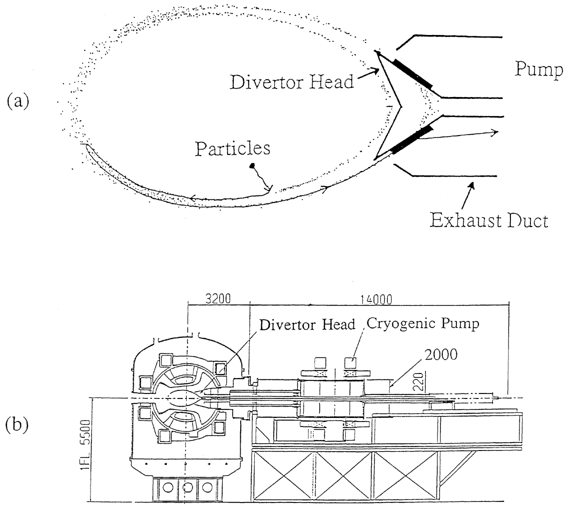

Fig. 2. (a) $m/n = 1/1$ island geometry for LID (b) Schematic view of the Local Island Divertor (LID).

particle flux cross the island separatrix by perpendicular diffusion and flow along the field lines toward the rear of the island, where target plates handle the heat load. The particles recycled there are pumped away very effectively by cryopumps. Its pumping efficiency is designed to be as high as 30%. Because of localization of the recycling, pumping is technically easy, but its power handling capability is limited to 4 MW.

One of the remarkable features of the island divertor configuration is a very sharp transition (within 2 mm in the radial direction) from the last closed magnetic surface (LCMS) to the open region in contrast to the helical divertor with a wide transition width (greater than 50 mm). This could be important for generating a so called H-mode thermal barrier (with typical radial width of 10–20 mm) located just inside the LCMS. In helical devices, H-mode [11] so far has been achieved only when rotational transform ($\iota/2\pi$) at the LCMS is 1.0 or 0.5 (the major rational surfaces) (in W7AS [12] and CHS [13]), but improvement of the energy confinement (τ_E) is very modest. With a closed divertor separatrix at $\iota/2\pi = 1.0$, a significant τ_E improvement might be achieved.

2.4. Simultaneous achievement of H-mode and radiative cooling (SHC operation)

A new boundary control scheme (SHC operation) has been proposed [14], which could allow simultaneous achievement of the H-mode type confinement improvement and edge radiative cooling with wide heat flux distribution. In our proposed configuration, the $m/n = 1/1$ island sharply separates the plasma confining region from the open 'ergodic' boundary (Fig. 3(a)). The connection length (between the point just outside of the LCMS and the divertor plate) is ~200 m (~8 × 2π *R*). It may be equivalent to tokamak poloidal divertors with long divertor channels (Fig. 3(b)). When collision with neutral particle is minimized by a baffle, high degree of openness (i.e., dominance of the parallel transport over the perpendicular transport) in the ergodic boundary makes the plasma pressure constant along the field line, which in turn separates low density plasma just outside

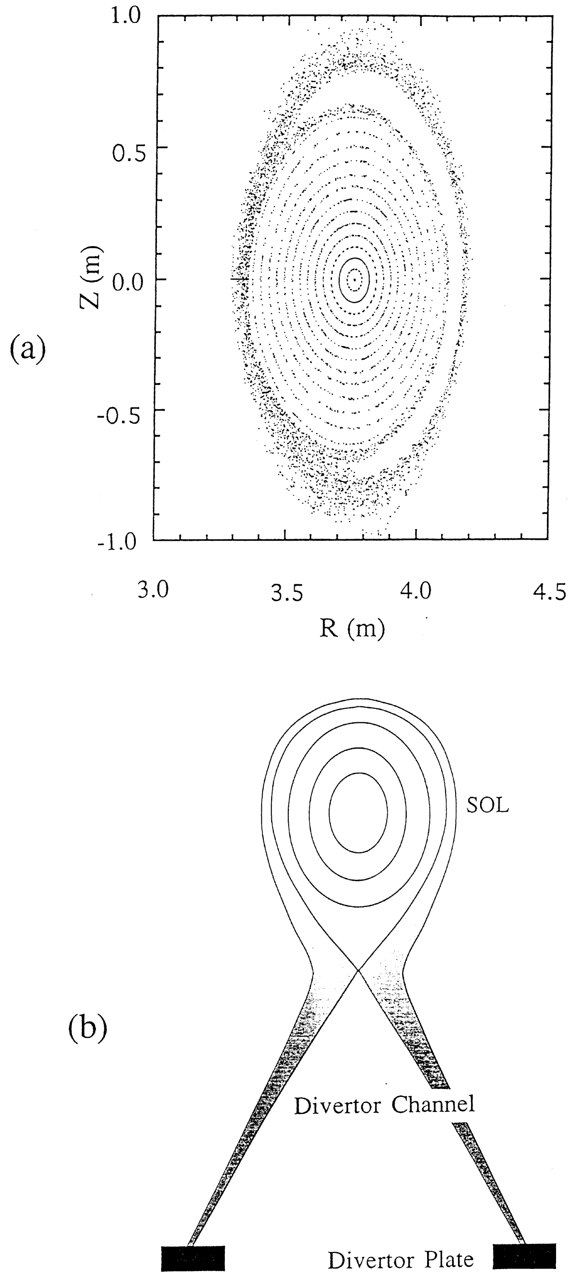

Fig. 3. The magnetic configuration for simultaneous achievement of H-mode and radiative cooling.

the plasma confining region (the key external condition for achieving a good H-mode discharge) from very high density, cold plasma near the wall (required for effective radiative cooling). In this approach, the magnetic configuration is the same as that of LID, but there is no target plate inserted in the island and pumping is not essential.

2.5. High temperature divertor plasma operation (HT-operation)

In the HT-operation [3], the edge temperature is raised up to a high value, several keV by efficient pumping and the resultant high edge temperature hopefully leads to enhancement in the energy confinement. The edge temperature is estimated to be $T_{\rm div} = W\eta/\gamma$ for an NBI heated and fueled discharge where W, η, γ are the beam energy, the pumping efficiency and the transmission coefficient at the sheath respectively, e.g., $T_{\rm div} \sim 4$ keV for $W \sim 180$ keV, $\eta \sim 0.2, \gamma \sim 10$. In this operation, a peaked density profile is maintained by a combination of deep fueling such as neutral beam or pellet injection and efficient particle pumping. Thus the diffusion coefficient (D) and hence the particle confinement becomes important in determining the energy confinement. This is a desirable feature for enhancing the energy confinement in LHD typed devices where high ripple induced electron heat loss tends to suppress the temperature gradient. However, the effective D is not high because the ions are well confined by $E \times B$ drift. Furthermore, the radial electric field in such a plasma regime is positive and hence neoclassical outward impurity pinch will prevent the impurity contamination [15].

The major uncertainty of this operation is unexpected interactions of high temperature edge plasma with the divertor plates. Recent results from low recycling tokamak divertor operations (JET [16], JT60-U [17]) are encouraging; fairly high edge ion temperature has been observed without accompanying any severe impurity problem.

The HT operation requires an efficient hydrogen pumping, motivating development of pumping schemes for LHD such as carbon sheet pumping [4] and membrane pumping [5]. In these pumping systems, thin pumping sheets cover a significant fraction of the vessel wall near the divertor region and absorb atomic hydrogen particles recycled from the divertor. For reactor application of the HT-operation, however, a new divertor magnetic geometry needs to be explored, which guides the outward flowing plasma to a very remote area with weak magnetic field, thereby allowing effective pumping and reliable heat removal even in the reactor environments. Taking advantage of a fact that relatively strong magnetic field extends beyond the helical coil cage in heliotron devices, such a divertor geometry can be configured [18].

3. LHD divertor experimental schedule

In the very early stage, we use the helical divertor with open geometry, characterized with a large volume of open edge region. It is ideal for radiative cooling operation. But this tends to raise the density at the LCMS, probably making formation of the H-mode barrier (located within a few cm from the LCMS) difficult. And thus it is not suitable in achieving an H-mode type confinement improvement.

Then we will install the LID in late 1999, which will allow low recycling discharges. This may lead to a better confinement regime. With the LID experiment (even though the power is limited to 4 MW) being done before the optimized helical divertor experiment, we will obtain critical information as to edge plasma behavior in LHD, particularly, physics insights into the relation between the edge plasma and the core plasma confinement and thus can optimize the design of the (upgrade) helical divertor. In addition, the LID discharge operation for an hour at low power (100–500 kW) will be a very effective discharge cleaning scheme.

Like the present tokamak approach, we will start to investigate the simultaneous attainment of the H-mode and radiative cooling (SHC operation) from late 1998. For LHD, an $m/n = 1/1$ magnetic island at the edge may play a key role, providing new important features: (i) the LCFS can be defined sharply, (ii) the cooling volume can be adjustable somewhat.

With pumping panels (carbon sheet or membrane pump) installed, the recycling can be minimized even in open helical divertor with high power (20 MW) handling capability. The edge temperature will be raised up to a few keV by NBI injection, thereby leading to enhancement in the energy confinement. However, the cost of such divertors is not small since the total length of the helical divertor leg is as long as 4×40 m. What is required is a simplified closed divertor configuration with high power handling and efficient pumping. The configuration depicted in Fig. 4, a candidate configuration for divertor upgrade during the second phase of the LHD experiment (2000 -) may satisfy such requirements. A set of divertor plate units (10 units in total) are located radially at slightly inside of the 'X-point' and poloidally at the small major radius side of the torus ($135° < \theta < 225°$). The majority of the outward flowing plasma particles are intersected by these divertor plates. The total area of the plate receiving the heat is expected to be around 2 m^2 and thus withstand 20 MW heating power. It extends helically only 2 m $\times$ 10 (units) instead of 40 m $\times$ 4 (legs) and thus its cost is an order of magnitude lower. Moreover, the shape of this type divertor unit is much simpler, the size of the unit is reasonably small, 0.3 m $\times$ 2.0 m and thus it can be designed to handle a steady state input power flux of 10

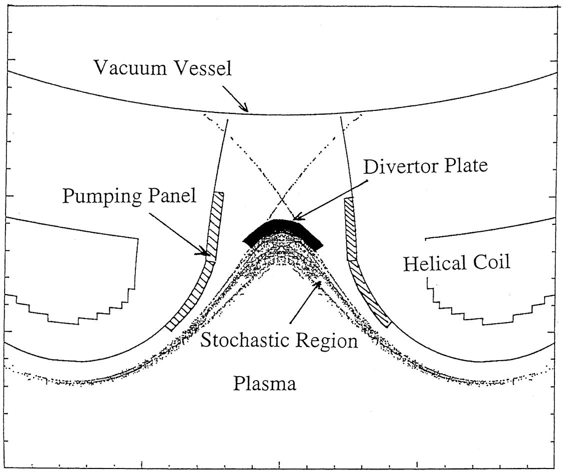

Fig. 4. Inboard helical divertor configuration. A candidate for the upgrade divertor configuration.

MWm^{-2}. For HT operation, pumping panels of the carbon sheet pump or the membrane pump are installed on the vacuum vessel wall near the divertor plate, as shown in Fig. 4. They will absorb the recycled hydrogen particles efficiently. For the SHC operation, localization of the recycling in the open region is an important requirement. It can be realized because the width of the open plasma region in front of the divertor plate is greater than 10 cm and the expected plasma density at LCMS is 2×10^{13} cm^{-3}.

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